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Overview of the US ITER Dual Coolant Lead Lithium (DCLL) Test Blanket Module Program*

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***Abstract*—With the US rejoining ITER, the US chamber technology community has resumed participation in discussion in the ITER Test Blanket Working Group (TBWG) and has proposed to develop, in collaboration with other parties, liquid and solid breeder blanket concepts to be tested in ITER. Presently, the US focus on the liquid breeder option is the dual coolant helium-cooled reduced activation ferritic steel structure with self-cooled Pb-17Li breeder (DCLL) that uses flow channel insert (FCI) as the MHD and thermal insulator. When projected for a reference tokamak power reactor design, it has the potential for a gross thermal efficiency of > 40%. The US is planning for an independent test blanket module (TBM) that will occupy half an ITER test port with corresponding supporting ancillary equipment. An initial design, testing strategy and corresponding test plan have been completed for the DCLL concept. The DCLL TBM conceptual design for the integrated testing phase, including the choice of configuration, relevant design analyses, ancillary equipment, testing strategy and corresponding test plan, have been prepared for the transition into the preliminary design phase.**

Keywords–blanket, Pb-17Li breeder, dual-coolant, helium-cooled, ITER-TBM

I. INTRODUCTION

With the U.S. rejoining ITER and in light of the new R&D results from the U.S. and world programs over the past decade, a study was initiated [1] to select two blanket options for the U.S. ITER Test Blanket Module (TBM). For consistency in blanket concept comparison, a set of DEMO-like parameters was selected with a maximum neutron wall loading of ~3 MW/m^2 and a maximum first wall (FW) surface heat flux of 0.5 MW/m^2. Three key conclusions were reached early in the study: (1) Selection between solid and liquid breeders cannot be made prior to fusion testing in ITER, and the U.S. should be engaged in testing both options at some level. (2) While some liquid breeder options do have potential for higher performance, they also have serious feasibility issues requiring more assessments. (3) In general, solid breeder is proposed by all parties, yet performance issues remain. During 2004, the U.S. Plasma Chamber Community team focused on the assessment of critical issues of liquid breeder concepts with strong participation from the U.S. materials, safety and plasma facing component programs. Examples of issues that were evaluated included magnetohydrodynamic (MHD) insulators, MHD effect on heat transfer, tritium permeation, corrosion, SiC_f/ SiC flow FCI viability and compatibility [1,2]. The dual coolant concept was selected as our primary liquid breeder option. A detailed description of this work is presented in the DCLL DDD report [3].

II. THE DCLL CONCEPT

As a liquid breeder blanket option, the dual coolant lead lithium (DCLL) blanket concept was selected as our primary concept for the TBM. With the use of reduced activation ferritic steel (FS) as the structural material, we are limited to a maximum steel structure temperature of <550°C. At the same time we have to remove the first wall heat flux, breed adequate tritium for the D-T fuel cycle and achieve high coolant outlet temperature for high power conversion efficiency. The basic approach of the DCLL concept is shown in Fig. 1 which shows the use of helium to cool the first wall and all FS structural elements, and the use of FCI element to perform the key functions of reducing the MHD effect of the circulating

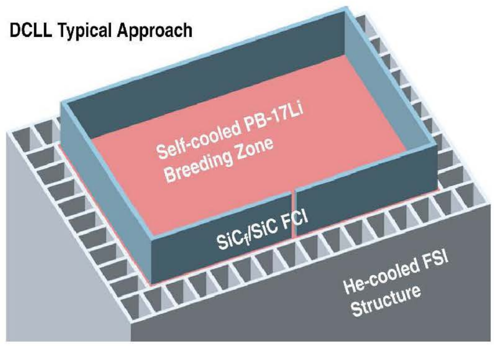

Figure 1. Dual coolant lead-lithium (DCLL) FW/blanket concept.

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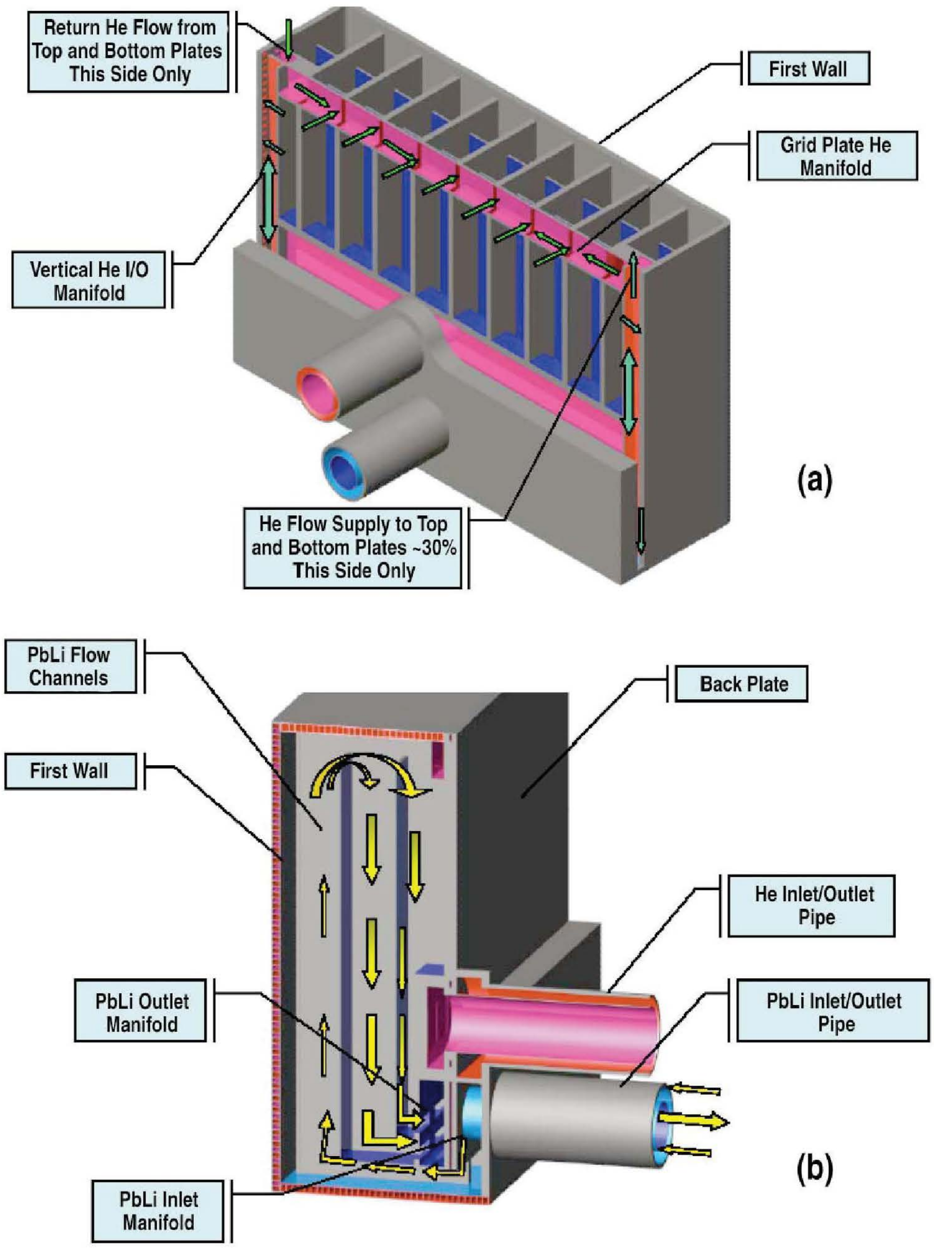

Figure 2. DCLL module for the reference reactor design. (a) Exposed view (red) He flow. (b) Pb-17Li flow (blue).

self-cooled Pb-17Li breeder in the magnetic field and thermally isolating the high temperature Pb-17Li in the main channel from the low temperature FS structure. The Pb-17Li is flowing in the larger channel at low velocity in order to achieve high outlet coolant temperature [4-6].

III. DCLL FOR REFERENCE REACTOR DESIGN

To select an ITER-TBM first wall and blanket design, we have to make sure that the concept can provide adequate performance when it is extrapolated to a high performance 2116 MW fusion tokamak reactor design [7]. We performed such an assessment with peak/average outboard neutron wall loading of 3.72/2.13 MW/m^2, and peak surface heat flux at outboard midplane of 0.5 MW/m^2. An outboard poloidally segmented blanket module for the reference reactor design is shown in Fig. 2. It has a two-pass poloidal Pb-17Li flow. The Pb-17Li inlet and outlet temperatures are 460°C and 700°C, respectively. The 8 MPa helium cools the first wall, and all the FS structure has helium inlet and outlet temperatures of 300°C and 480°C, respectively. Concentric pipes are used for both Pb-17Li and helium coolants circulating in and out of the blanket module. Neutronics calculations have been performed to determine the important nuclear performance parameters. The overall tritium breeding ratio is estimated to be 1.15 with Li6 enrichment at 90%, excluding any breeding in the divertor region. With the use of multiple-reheat Brayton closed cycle gas turbine power conversion system [4,6-10] a gross power conversion efficiency of >40% can be projected.

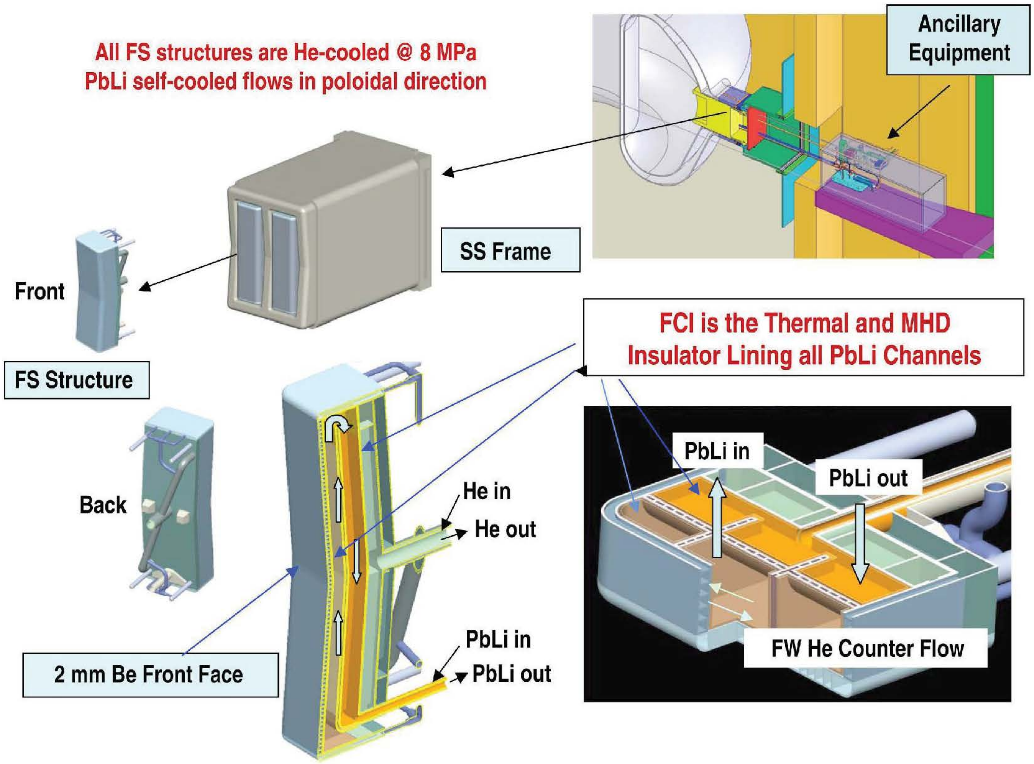

Figure 3. US DCLL TBM module design.

IV. DCLL ITER-TBM DESIGN

The DCLL ITER-TBM design is shown in Fig. 3. We have selected a similar blanket configuration as for the reference tokamak reactor design shown in Sec. III. Design parameters for the ITER-TBM are presented in Table I.

TABLE I. ITER-TBM DESIGN PARAMETERS

Fusion power	500 MW
Design heat flux	0.5 MW/m^2 for 10% of the area 0.3 MW/m^2 for 90% of the area
Design neutron wall loading	0.78 MW/m^2
Disruption heat load	0.55 MJ/m^2
Plasma burn time	400 s
Time between shots	2000 s
Duty factor	0.25

Figure 3 shows the counterflow helium-cooled first wall design, the poloidal flow of the Pb-17Li and the insertion of the DCLL into the frame of the ITER test port. The relative location of the test port at the mid-plane of the ITER chamber and the corresponding transporter behind the test module are also shown. The module has to be designed to withstand a helium pressure of 8 MPa under loss of coolant accident conditions. As shown, for the safety concern in the generation of hydrogen due to the potential Pb-17Li interaction with water under accident conditions, the volume of Pb-17Li is limited to 0.27 m^3, leading to the TBM radial thickness of 0.413 m. The helium coolant is designed to pass the first wall five times before leaving the first wall panel. In order to maintain the first wall maximum allowable temperature of <550°C, the first wall cooling channel surface on the plasma side will have to be roughened in order to enhance the heat transfer coefficient. The helium loop total pressure drop is 0.81 MPa, which is high but acceptable for a testing device.

Initial 1-D neutronics calculations for the TBM were performed. With the limited geometry of the TBM in the ITER machine, the local tritium breeding ratio is only 0.741 with 0.413 m module depth. Tritium is generated at the rate of 3.2x10^{17} atoms/s in the TBM. Nuclear heating in different components of the TBM was evaluated and used for the thermal-hydraulics assessment.

A key element in the DCLL is the SiC-composite FCI. With the possibility of tailoring the electrical and thermal properties of the SiC-composite material by controlling the manufacturing process of the composite, we performed a detailed MHD analysis to determine the design window of critical FCI material properties [11]. Flow equations coupled with the induction equation were solved for the FCI design assuming fully developed flow conditions using MHD code specially developed for such flows [12]. For the TBM, the temperature profile of the Pb-17Li was calculated The maximum temperature can reach 675°C with a SiC_f/SiC effective thermal conductivity of 15 W/m-K. For the reference reactor design we also found that the suitable range of electrical and thermal conductivity for the SiC-composite are $\sigma = 20 - 50\, 1/\Omega - m$, and k = 2-5 W/m-k, respectively. The total coolant loop MHD pressure drop at an electrical conductivity of 20 S/m, including the poloidal channel flows, the inlet and outlet manifolds and the concentric tube, is estimated to be 0.3 to 0.44 MPa.

Pb-17Li has low tritium solubility; therefore the tritium mobility in the DCLL blanket system is high. For the DCLL design it becomes critical to have an efficient method of extracting tritium from the Pb-17Li in order to minimize the amount of tritium release to the public. For the reference reactor design, the outlet Pb-17Li has a maximum temperature of 700°C. The critical element of this loop is the proposed vacuum permeator tritium extraction system, with the Pb-17Li flowing in thin wall tubes and the tritium being extracted via high vacuum outside of the tube bank. Single tube vacuum permeator performance and the corresponding tritium loop equilibrium calculation, utilizing the tritium migration analysis program (TMAP) [13], were performed. Results show that with a Nb tube (1 cm diameter and wall thickness of 0.5 mm) and length of 4 m, an exit tritium pressure of 0.2 Pa can be achieved for a mass transfer coefficient of 5.6 m/s. When the power conversion loop can be divided into four sectors, an acceptable tritium inventory of less than 100 gm in the heat exchanger system per quadrant of the 2116 MW fusion reference tokamak reactor could be achieved. These results show the potential feasibility of the vacuum permeator extraction approach, but many basic R&D issues remain, including the control of the reaction between Nb and oxygen impurities in the vacuum side. Similarly, we will have to assess the suitable heat exchanger material for high temperature Pb-17Li and helium. Nb and other refractory alloys with their allowable high operation temperature and potential compatibility with Pb-17Li are candidate materials to be evaluated.

V. ANCILLARY CIRCUITS AND EQUIPMENT

The DCLL ancillary equipment assessment was based on the requirements of handling different operation/testing scenarios of the DCLL including the maximum FW heat flux and maximum outlet temperature of the breeder coolant up to 650°C as well as low temperature operation during the early phases of ITER testing. From the DCLL TBM we have two coolant loops. The first one is the FW and structure helium coolant loop, which was designed to carry 54% of the total blanket power at the maximum ITER operation level during the D-T phase. Dedicated helium piping is then designed between the TBM and the helium/water heat exchanger at the Torus Cooling Water System (TCWS) vault. The second one is the liquid breeder loop, which was designed to carry 46% of the blanket energy during the D-T phase of ITER operation. In addition to the Pb-17Li breeder loop and corresponding equipment located on the transporter as shown in Fig. 4, a helium intermediate heat removal loop was designed between the transporter and the TCWS water cooling system. This is to avoid the extension of the Pb-17Li piping to the TCWS vault, which is ~70m away from the TBM test port. Corresponding helium piping and equipment were also designed. The ancillary equipment for the primary and intermediate helium systems in the TCWS vault is shown in Fig. 5.

The liquid breeder-to-helium heat exchanger is located close to the test module on the transporter to minimize the amount of liquid breeder and the corresponding loss of tritium to the surroundings. To avoid the use of advanced materials for the handling of high temperature Pb-17Li, a bypass loop system as shown in Fig. 6 is selected. Hot Pb-17Li returning from the TBM is mixed with the bypassed cold Pb-17Li at the bypass section, resulting in only a warm stream limited to 470°C going to the tritium extraction and heat exchanger systems. In this way, the high temperature features of the TBM, especially the function of the SiC_f/SiC FCI as a thermal insulator at high temperature, can be tested in ITER without requiring high temperature materials in the tritium extraction and heat exchanger systems.

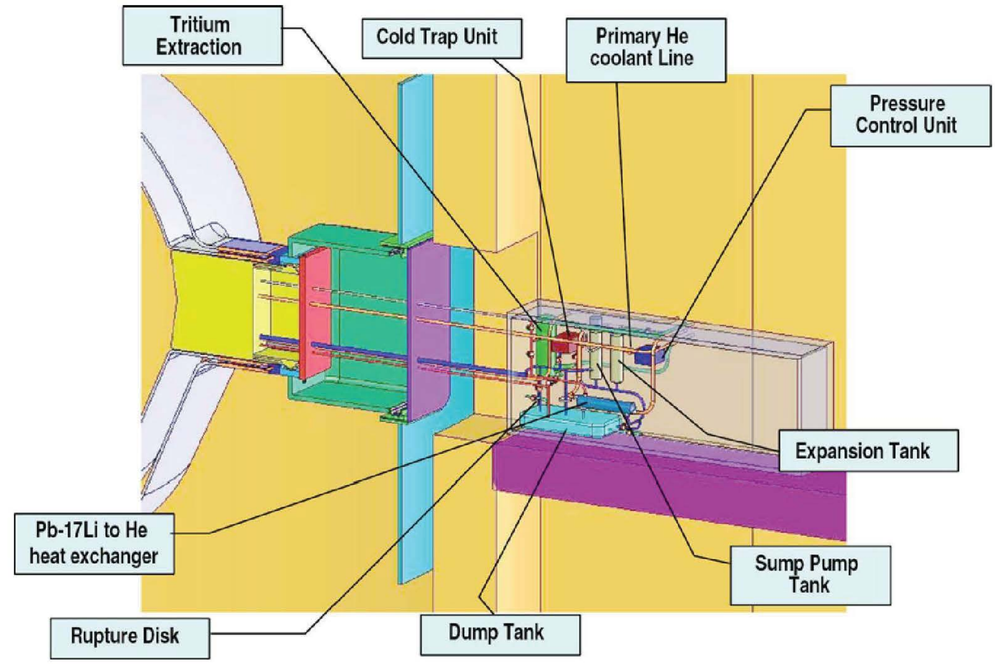

Figure 4. Ancillary equipment of the PbLi circuit in the transporter.

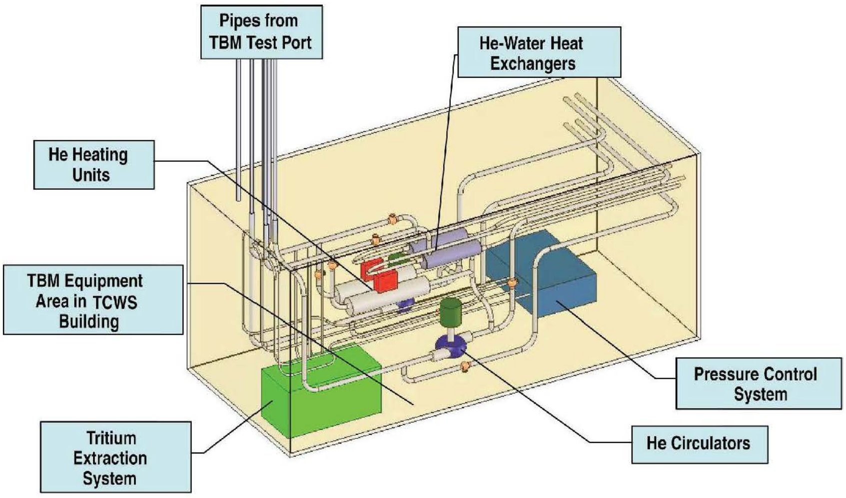

Figure 5. DCLL primary and intermediate helium coolant components in TCWS.

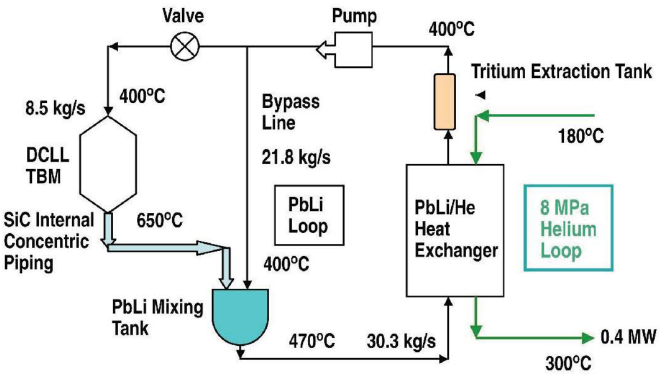

Figure 6. DCLL TBM bypass loop schematic.

VI. DCLL TBM Test Plan

The U.S. strategy for ITER testing of the DCLL concept is flexible and still evolving. The test plan must remain flexible in order to respond to technical issues that are revealed only during testing, as well as budget issues that must also be accommodated. The baseline assumption underlying the current planning and TBWG documentation is to design:

- for a series of vertical half-port DCLL TBMs,
- with dedicated ancillary equipment systems in transporter casks behind the bioshield and in space in the TCWS building,
- during the period of the first 10 years of ITER operation.

The strategy for ITER testing progresses from basic structural and MHD performance tests to more integrated module tests as a function of the ITER operational phases during the first 10 years. It should be noted that this test program is developed assuming successful testing in previous phases. A series of four test modules can be characterized as:

EM/S TBM: The first test module is an Electromagnetic/ Structural (EM/S) module designed to withstand and measure EM forces and the mechanical response of the TBM structure to such loads during ITER hydrogen phase operations including: chamber conditioning, startup, shutdown, normal discharges and transient effects including ELMs and disruption.

NT-TBM: Following the EM/S TBM a Neutronics TBM (NT TBM) will be tested during the D-D phase and possibly the very beginning of the low duty cycle D-T phases. These neutronics tests can be performed on a similar "look-alike" structure as the EM/S TBM.

T/M TBM: At the beginning of the low duty cycle D-T phase a Thermofluid/MHD (T/M) TBM is planned. The strategy for the T/M TBM is to allow testing of a variety of flow channel insert (FCI) design, geometries and integrated functions.

I TBM: During the high duty cycle DT phase an Integrated (I) TBM is planned where the longer term operation of the system is explored including some accumulation of radiation damage in the FCI, and tritium and transmutation products in the Pb-17Li.

Detail description of the test plan can be found in the DDD report [3].

VII. Conclusions

We have completed our assessment and selection of the DCLL as the liquid breeder first wall and blanket concept for the ITER-TBM. For the reference tokamak power reactor design, this blanket design concept has the potential of satisfying the design limits of FS while allowing the feasibility of having high Pb-17Li outlet temperature of 700°C. Using the Brayton cycle power conversion system, the projected gross thermal efficiency is >40%. We have begun the DCLL ITER-TBM design including the assessment of critical issues. Some of these include the first wall design, the assessment of MHD effects with the SiC-composite FCI design, and the extraction and control of the bred tritium from the Pb-17Li breeder. The suitable range of electrical and thermal conductivity for the SiC-composite are σ=20-50 1/Ω-m, and k=2-5 W/m-k, respectively. The vacuum permeator tritium extraction system seems to be credible for tritium extraction and control, but basic R&D issues need to be addressed. We have completed the scoping design of the ancillary equipment supporting the DCLL ITER-TBM test program. The ancillary equipment circuit system consists of the primary helium circuit, the Pb-17Li circuit and the intermediate helium circuit. A corresponding test plan for the first 10 years of ITER operation has been proposed, and we are ready to progress to the next stage of preliminary design.

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